NON-PUBLIC?: N
ACCESSION #: 8712040171

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Virgil C. Summer Nuclear Station PAGE: 1 of 5

DOCKET NUMBER: 05000395

TITLE: Reactor Trip Resulting From Power Loss to Process Rack
Panel XPN 7008
EVENT DATE: 10/29/87 LER #: 87-027-00 REPORT DATE: 11/25/87

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: W. R. Higgins - Associate Manager, Regulatory Compliance
TELEPHONE #: 803-345-4042

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JC COMPONENT: PL MANUFACTURER: W120
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On October 29, 1987, at 0313 hours with the plant at 100% power, both the primary and backup power supplies to one of the Westinghouse 7300 system process rack panels failed. This resulted in a loss of various instrumentation and control functions including feedwater controls, pressurizer pressure and level controls, and steam dump controls. Manual control of the feedwater system was attempted, but due to the nature of the control failures, recovery was not possible. The plant tripped on low steam generator "C" level coincident with steam flow/feed flow mismatch. A pressurizer power operated relief valve lifted, the steamline power operated relief valves lifted, and the motor driven and turbine driven emergency feedwater pumps started. Within approximately 13 minutes I&C personnel had restored power to the panel and control systems returned to normal. The plant, utilizing the reestablished control functions, recovered normally from the trip.

The cause of the event was attributed to a failed capacitor on a steam dump control signal converter card in one of the process rack panels. This

capacitor shorted to ground causing the breakers to both the primary and backup power supplies for the process rack panel to trip.

Westinghouse, the supplier of the panel, has been requested to provide justification for the acceptability of the design. In addition, South Carolina Electric & Gas Company (SCE&G) is investigating a design change to minimize the possibility of a recurrence.

(End of Abstract)

TEXT: PAGE: 2 of 5

PLANT IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION:

Process Rack - Control Group 4 of Westinghouse 7300 System - XPN 7008 (EIIS:JH)
Feedwater System - (EIIS:SJ)
Pressurizer - (EIIS:AB)
Steam Dumps - (EIIS:SB)

Feedwater Regulating Valves - (EIIS:SJ)
Steam Generator - (EIIS:SB)
Main Feedwater Pumps - (EIIS:SJ)
Emergency Feedwater Pumps - (EIIS:BA)
Pressurizer Power Operated Relief Valves (EIIS:AB)
Steamline Power Operated Relief Valves (EIIS:SB)
Refueling Water Storage Tank - (EIIS:BQ)
Volume Control Tank - (EIIS:CB)
Reactor Coolant System - (EIIS:AB)
Charging Pumps - (EIIS:BQ)

IDENTIFICATION OF EVENT:

Reactor Trip resulting from a power loss to process cabinet XPN-7008.

PREVIOUS SIMILAR EVENT:

None

EVENT DATE: 10/29/87 at 0313 hours

REPORT DATE: 11/25/87

This report was initiated by Off-Normal Occurrence Number 87-099.

CONDITIONS PRIOR TO EVENT:

Mode 1 - Reactor 100% Power

TEXT: PAGE: 3 of 5

DESCRIPTION OF EVENT:

On October 29, 1987, at 0313 hours with the plant at 100% power, Channel IV process rack and control group cabinet XPN 7008 primary and backup power supply breakers tripped. This resulted in various instrumentation and control function failures including several feedwater system controls, pressurizer pressure and level controls and condenser steam dump controls. The first indication of the problem was the steam generator feedwater flow/steam flow mismatch alarm. Operations personnel immediately attempted to gain control of the feedwater system. Manual control of the feedwater regulating valves was taken to open the valves; however, no response was seen. The reactor tripped on the low steam generator "C" level coincident with steam flow/feed flow mismatch. All three steam generator levels decreased to approximately 8% resulting in an Emergency Feedwater actuation of the two motor driven and the turbine driven Emergency Feedwater pumps. As a result of the loss of controls, the following control functions failed: 1) the main feedwater pumps lost speed control; 2) the steam dump valves did not operate and 3) pressurizer spray valves would not actuate. A manual trip of the Main Feedwater Pumps from the control board was initiated and the "A" and "B" pumps tripped. The "C" pump did not trip; an auxiliary operator tripped the pump locally. The pressurizer power operated relief valve opened as designed at 2335 psi to relieve primary pressure and remained open for a brief period of time (approximately four seconds). Steam pressure increased, the steamline power operated relief valves opened and manual control was taken to control secondary pressure. The maximum steam pressure reached was 1150 psig and the power operated relief valves were activated for approximately six minutes.

Due to the cooldown on the primary side, the volume control tank (VCT) level decreased and a letdown isolation occurred. To prevent a safety injection from occurring, the operators aligned charging pump suction to the Refueling Water Storage Tank and started another pump in order to maintain reactor coolant system inventory. When VCT level recovered, the charging pumps were realigned to take suction from the VCT. During the transient, the pressurizer pressure ranged from 2335 psig to 1940 psig with a minimum level of 17 percent. Tavg ranged from 591 degrees F to 535 degrees F.

Approximately 13 minutes after the trip, the I&C supervisor had reached XPN 7008 and found the output breakers from the primary and backup power supplies tripped. The breakers were closed and all control functions were recovered. The plant recovery to stable conditions was resumed using normal control functions.

CAUSE OF EVENT:

The cause of this event was the failure of a capacitor on a signal converter card for the steam dumps located in XPN 7008. This capacitor failure shorted both the primary and backup power supplies and caused both magnetic breakers to trip instantaneously. Subsequent to shorting, the capacitor burned and eventually created an open circuit. This enabled the power supply breakers to be closed.

ANALYSIS OF EVENT:

This report is being submitted pursuant to the requirement of 10CFR50.73(a)(2)(iv). Notification to the NRC Operations Center via the Emergency Notification System was also made pursuant to the requirements of 10CFR50.72(b)(2)(ii).

Activation of the steamline power operated relief valves and the turbine driven emergency feedwater pumps resulted in a radioactive release of 8.14E-4 percent of the Technical Specifications limit due to minor primary to secondary leakage in the steam generators.

There were no safety consequences associated with this event since the reactor protection system functioned as designed and maintained the plant in a safe condition throughout the transient. In addition, prompt operator actions in recognizing, evaluating and controlling the transient minimized the effects of the event.

I&C personnel had reached the affected panel, found the tripped breaker, and reset the breaker at the direction of operations personnel within 13 minutes from the initiation of the event.

Westinghouse was contacted to determine why the design allowed one failed capacitor to inhibit both power supplies. Westinghouse explained that the normal failure mode of the capacitor was to open and not short; therefore, the possibility of the capacitor shorting was not assumed as part of the design considerations.

IMMEDIATE CORRECTIVE ACTION:

Operations personnel quickly and effectively analyzed the event. I&C was notified and sent to the proper location to reset the tripped breaker thereby restoring power to the control functions within 13 minutes. In addition, the control room personnel anticipated plant responses and took actions to start an additional charging pump and switch suction to the Refueling Water Storage Tank when Volume Control Tank levels decreased. These actions prevented a possible safety injection from occurring.

A sticking solenoid was determined to be the cause of the "C" Feedwater Pump failing to trip. The solenoid was exercised several times and the problem could not be repeated. It was subsequently returned to service.

The failed signal converter card in XPN 7008 was identified and replaced.

ADDITIONAL CORRECTIVE ACTION:

A Management Review Board Meeting, chaired by the Vice President, Nuclear Operations, convened to investigate the details of the event and identify corrective actions necessary to prevent recurrence.

Westinghouse has been requested to provide the design intent of the circuit card and a justification for the acceptability of the design.

An investigation has been initiated to determine if a design change is feasible in order to prevent an event of this type from occurring again.

SCE&G will have completed the above items by February 29, 1988. Any design changes will be initiated and sent through the normal screening and prioritization committees to obtain implementation schedules.

10CFR50.73

SCE&G South Carolina Electric & Gas Company Dan A. Nauman
P.O. Box 764 Vice President
Columbia, SC 29218 Nuclear Operations
(803)748-3513

November 25, 1987

Document Control Desk

U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
LER 87-027

Gentlemen:

Attached is Licensee Event Report No. 87-027 for the Virgil C. Summer Nuclear Station. This report is submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv).

Should there be any questions, please call us at your convenience.

Very truly yours,
/s/ Dan A. Nauman
Dan A. Nauman

AMM/DAN:bjh
Attachment
pc: O. W. Dixon, Jr./T.C. Nichols, Jr. C. L. Ligon (NSRC)
E.C. Roberts R. M. Campbell, Jr.
O. S. Bradham K. E. Nodland
D. R. Moore J. C. Snelson
W. A. Williams, Jr. G. O. Percival
J. Nelson Grace R. L. Prevatte
Group Managers J. B. Knotts, Jr.
W. R. Baehr INPO Records Center
W. R. Higgins Marsh & McLennan
R. B. Clary ANI Library
G. G. Soult RTS
C. A. Price NPCF
W. T. Frady L. M. Montondo
T. L. Matlosz File

*** END OF DOCUMENT ***